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Abstract

IRRADIATION BEHAVIOR OF CERAMIC FUELS

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A review is presented of the extensive investigations made since 1958 of various ceramic compounds of uranium, plutonium, and thorium for use as reactor fuel materials. The ceramic fuel materials most extensively investigated are the uranium oxides, followed by the carbides and to a lesser extent the nitrides and the silicides. This paper reviews the effects of radiation at various power levels on the properties and performance characteristics of ceramic fuel materials.

The ease of fabricability, stability under irradiation, and compatibility with high temperature water coolant has led to the successful use of the oxides, principally UO_2 and to a lesser extent ThO_2-UO_2 and PuO_2-UO_2, as fuel materials for water cooled power reactor applications. Other oxides being investigated include ZrO_2 mixtures with UO_2 and PuO_2, BeO-UO_2, UO_2-CaO-ZrO_2, UO_2-CeO_2 and Mgo-PuO_2.

UO_2 pellet and compacted powder fuel rods have been operated at thermal ratings high enough to result in melting of the central portion of the fuel. Observations of volume swelling of fuel rods operating with molten UO_2 and the use of cored pellets or lower density compacted UO_2, to eliminate volume swelling are discussed.

The results of UO_2 central melting experiments which have led to improved correlations between fuel rod thermal ratings, UO_2 thermal conductivity, changes in UO_2 structure, and identification of molten zone are discussed. Data are presented on the thermal conductivity of polycrystalline as-sintered

UO_2 at elevated temperatures and on changes in thermal conductivity of UO_2 with burnup.

Quantitative data are presented on the volume swelling rate of UO_2 as a function of burnup, UO_2 density, and operating temperature. Other significant data on UO_2 fuel include fission gas release measurements on fuel operating up to and above the melting temperature, observation of metallic inclusions of uranium in UO_2 irradiated at high temperatures, and low-temperature sintering of compacted powder UO_2 during irradiation.

The results of extensive irradiation tests of other oxide fuel materials such as ThO_2-UO_2, PuO_2-UO_2, and ZrO_2-UO_2 are presented. Results of recent tests of PuO_2-UO_2 fuel irradiated to 20 x 10^{20} fissions/cc indicate that no radial migration of Pu, Zr-95 or Ce-144 occurred. ZrO_2-UO_2 was found to be a stable material for use in plate-type elements. The swelling rate of ZrO_2-UO_2 with burnup was measured to be the same as that of UO_2.

The higher thermal conductivity and metal atom density of the carbides relative to the oxides has provided an incentive for the investigation of such ceramic fuel compounds as UC, PuC and UC-PuC mixtures. The most extensively studied of the carbides is UC. Data on the relative stabilities of hypo-stoichiometric and hyperstoichiometric UC as a function of operating temperature and irradiation exposure are presented. Peformance data for PuC and UC-20 w/o PuC, irradiated to burnups of 7.1 x 10^{20} fissions/cc, are discussed.

Uranium monoitride and uranium monosulfide have been investigated as potentially better ceramic fuel materials than UO_2 and UC. The results of irradiation tests of UN and US are reviewed.